July 28, 2006

EA-03-009

Mr. Paul A. Harden
Site Vice President
Nuclear Management Company, LLC
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT
NRC INTEGRATED INSPECTION REPORT 05000255/2006004

Dear Mr. Harden:

On June 30, 2006, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Plant. The enclosed report documents the inspection findings which were discussed on July 10, 2006, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, four NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because the violations were of very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the Enforcement Policy. If you contest the subject or severity of a NCV, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Palisades facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Christine A. Lipa, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket No. 50-255
License No. DPR-20

Enclosure: Inspection Report 05000255/2006004
w/Attachment: Supplemental Information

cc w/encl: J. Cowan, Executive Vice President
and Chief Nuclear Officer
R. Fenech, Senior Vice President, Nuclear
Fossil and Hydro Operations
D. Cooper, Senior Vice President - Group Operations
L. Lahti, Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel and Secretary
A. Udrys, Esquire, Consumers Energy Company
S. Wawro, Director of Nuclear Assets, Consumers Energy Company
Supervisor, Covert Township
Office of the Governor
State Liaison Office, State of Michigan
L. Brandon, Michigan Department of Environmental Quality -
Waste and Hazardous Materials Division

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2006004
Licensee:	Nuclear Management Company, LLC
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	April 1 through June 30, 2006
Inspectors:	J. Ellegood, Senior Resident Inspector J. Giessner, Resident Inspector A. Barker, Acting Resident Inspector R. Lerch, Project Engineer J. House, Senior Radiation Specialist J. Cassidy, Radiation Specialist F. Ramirez, Reactor Engineer G. Wright, Project Engineer T. Bilik, Reactor Inspector
Approved by:	C. Lipa, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000255/2006004; 04/01/2006 - 06/30/2006; Palisades Nuclear Plant; Operator Performance During Non-routine Evolutions and Events; Refueling Outage Activities; Event Followup.

This report covers a 3-month period of baseline inspections. The inspections were conducted by Region III inspectors and resident inspectors. This report includes four green findings, all of which were NCVs. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealed Finding

Cornerstone: Initiating Events

- Green. A self-revealed Non-Cited Violation (NCV) of Technical Specification (TS) 5.4 of very low safety significance was identified on May 11, 2006, when abnormal flux distribution prevented the licensee from continuing power ascension. The licensee determined a rod was not latched. The licensee violated TS 5.4, "Procedures," during performance of rod latching activities. The licensee's procedures were not adequate to latch the rod, and ensure the rod was latched prior to power operations. The licensee entered the item into the corrective action program. This finding also affected the cross cutting aspect of human performance. Immediate corrective actions included shutting down the reactor and latching the rod.

  The inspectors determined the finding is more than minor since the finding affected cornerstone objectives for both initiating events and mitigating systems. Specifically, the inserted rod reduced available shutdown reactivity and shifted core flux to reduce margin to thermal limits. The finding was of very low safety significance because power remained very low, less than 25 percent, core thermal limits were not violated, and adequate shutdown margin existed. (Section 1R20)

- Green. A self-revealed NCV of TS 5.4 occurred on April 22, 2006, when the polar crane bridge struck and severely damaged the jib crane. The licensee violated TS 5.4 for failing to have adequate procedures in place during maintenance that could affect safety-related equipment. The licensee entered the finding into their corrective action program. Immediate corrective actions included safely lowering attached loads, removing the crane from service, and inspecting affected equipment. This finding affected the cross cutting aspect of human performance.

  The inspectors determined the finding is more than minor since the finding could reasonably be seen as a precursor to more significant events. Specifically, failure to control load movements could result in heavy load drops. The finding is of very low safety significance since no loads were dropped and the damage that did occur did not affect inservice safety systems. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 for improperly removing commitments to maintain a keep warm pump from the Final Safety Analysis Report (FSAR). The licensee had committed to maintaining this pump in lieu of inspections of the intake structure. The licensee entered the item in the corrective action program and performed immediate corrective actions, including inspections of the intake structure.

  The inspectors concluded this finding is more than minor since it impacted the NRC's ability to perform its regulatory function. Specifically, the licensee changed the FSAR in a manner that required prior NRC approval. The finding is a Severity Level IV violation consistent with the NRC Enforcement Policy. (Section 4OA5)

Cornerstone: Barrier Integrity

- Green. A self-revealed NCV of Criterion XVI was identified when damage to a fuel pin was found. The finding of very low safety significance (Green) occurred because the licensee failed to assure adequate corrective actions were implemented to prevent recurrence of fuel cladding damage to a fuel assembly. This finding represented an NCV of 10 CFR 50, Appendix B, Criterion XVI, in that the appropriate actions were not in place for a significant condition adverse to quality. The licensee entered the item into the corrective action program. Immediate corrective action included changing the core design and replacing susceptible fuel rods with stainless steel pins.

  The inspectors determined that the finding is more than minor since the finding impacted the Barrier Integrity cornerstone objective of fuel clad integrity. Specifically, the clad on one fuel element had fretted away exposing the fuel plenum and plenum spring. The finding is of very low safety significance because only the fuel barrier was affected and plant TSs were not exceeded for fission product activity in the coolant. (Section 1R20)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant operated at or near full Rated Thermal Power during the inspection period with the following exceptions:

- The plant began the inspection period shutdown for the 18th refueling outage.
- On May 10, 2006, the plant was synchronized to the grid and raised power to 24 percent.
- On May 11, 2006, the licensee shut down the plant when an inserted control rod was identified.
- On May 16, 2006, the plant was synchronized to the grid and achieved 100 percent power on May 24, 2006.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Warm Weather Preparations

a. Inspection Scope

The inspectors verified that licensee personnel had completed their warm weather preparations as specified in Standard Operating Procedure 23, Attachment 10, "Warm Weather Checklist." The inspectors utilized this checklist to verify that the prescribed actions had been completed for safety-related equipment such as the emergency diesel generators and service water intake structure. The inspectors also reviewed selected condition reports related to warm weather preparation and verified that identified problems were entered into the corrective action program with the appropriate significance characterization, and that planned and completed corrective actions were appropriate.

The review as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors completed four equipment alignment inspection samples by performing partial walkdowns on the following risk-significant plant equipment:

- right train high pressure safety injection and containment spray alignment during isolation of one shutdown cooling heat exchanger;
- component cooling water system alignment during component maintenance in shutdown plant conditions;
- left train service water system alignment following system maintenance; and
- right train control room ventilation alignment during left train control room ventilation outage.

During the walkdowns, the inspectors verified that power was available, that accessible equipment and components were appropriately aligned, and that no open work orders for known equipment deficiencies existed which would impact system availability.

The inspectors also reviewed selected condition reports related to equipment alignment problems and verified that identified problems were entered into the corrective action program with the appropriate significance characterization and that planned and completed corrective actions were appropriate and implemented as scheduled. The documents reviewed during this inspection are listed in the attachment.

b. Findings

Introduction

The inspectors identified an issue associated with the licensee's removal from service of one shutdown cooling heat exchanger (SDCHX) with the refueling cavity drained and the primary coolant system vented. The licensee restored the lineup to two heat exchangers after maintenance. This is an unresolved item pending completion of an assessment by NRR in accordance with the Task Interface Agreement process.

Description

During a review of system lineups for replacement of CV-3070, left train high pressure safety injection subcooling valve, the inspectors noted that one SDCHX was removed from service and tagged out as part of the isolation for the work. The inspectors questioned the operating shift and operations management as to why this was an acceptable lineup for shutdown cooling. Technical Specification (TS) 3.9.5 requires two SDC trains to be operable when in mode 6 with cavity level less than 647 feet. The inspectors assessed two SDCHXs as needed for two SDC trains based on the TS limiting condition for operation (LCO) and the basis which defined an operable train consisting "of SDC pump, a heat exchanger, valves, piping, instruments and controls to ensure an Operable flowpath and to determine primary coolant system (PCS) temperature." The licensee contends that since Palisades had not been designed for a single passive failure and the heat exchangers are one unit, then a single sub-unit can be isolated provided primary temperature can be maintained. The licensee and NRC management discussed the issue and the licensee stated the existing configuration complied with their license. Further, the licensee prepared a TS basis change, using the 10 CFR 50.59 process to clarify the licensing basis. The TS basis change indicated that both heat exchangers operate as a single heat exchanger with two partial capacity units. The licensee determined this TS basis change was an enhancement and the 10 CFR 50.59 screen concluded a 10 CFR 50.59 evaluation was not required. The inspectors reviewed the screen and supporting documents, but did not agree with the licensee's conclusion that the change was only an enhancement not requiring a 10 CFR 50.59 evaluation. Without performing a 10 CFR 50.59 evaluation, the inspectors concluded that prior NRC approval may be required. This issue involves several complex parts and requires a response from NRR in accordance with the Task Interface Agreement process. Pending further NRR review, this issue is tracked as Unresolved Item (URI) 05000255/2006004-01. Since the licensee is currently not in this alignment and the risk of possible failures to the SDCHX is unlikely, there is no current safety issue.

1R05 Fire Protection

.1 Fire Area Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors completed six fire protection inspection samples by touring the following areas in which a fire could affect safety-related equipment:

- Component Cooling Water Rooms (Fire Area 16)
- Hot Work MV-SW-138 piping replacement in West Safeguards (Fire Area 28)
- East and West Engineering Safeguards (Fire Area 10)
- Cable Spreading Room (Fire Area 2)
- 1-D Switchgear Room (Fire Area 3)
- B Control Room HVAC equipment room (Fire Area 29)

The inspectors verified that transient combustibles and ignition sources were appropriately controlled, and that the installed fire protection equipment in the fire areas corresponded with the equipment which was referenced in the Updated Final Safety Analysis Report, Section 9.6, "Fire Protection." The inspectors also assessed the material condition of fire suppression systems, manual fire fighting equipment, smoke detection systems, fire barriers and emergency lighting units. For selected areas, the inspectors reviewed documentation for completed surveillances to verify that fire protection equipment and fire barriers were tested as required to ensure availability.

The inspectors reviewed selected condition reports associated with fire protection to verify that identified problems were entered into the corrective action program with the appropriate significance characterization. The inspectors also verified that planned and completed corrective actions were appropriate. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From April 10, 2006, through April 21, 2006, the inspectors conducted a review of the implementation of the licensee's Risk-Informed Inservice Inspection Program (RI-ISI) program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries. The inspectors selected the licensee's RI-ISI program components and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the on-site inspection period.

The inspectors observed the following three types of nondestructive examination activities to evaluate compliance with the ASME Code Section XI and Section V requirements, and to verify that indications and defects (if present) were dispositioned, in accordance with the ASME Code Section XI requirements:

- Visual Examination (VT) of eighteen Reactor Pressure Vessel Closure Head (RPVCH) closure stud washers;
- Magnetic Particle Examination of a RPVCH closure stud;
- Ultrasonic Examination (UT) of pressurizer weld PZR-017, weld number 22, a pipe-to-elbow weld; and
- UT of heater drain line weld HED-001, weld number 5.

There were no examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service.

The inspectors reviewed pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous refueling outage to determine if the welding acceptance and preservice examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed welds associated with the following work activities:

- Replacement (welding) of ISI class 1 check valve CK-CVC2114, a charging line loop-1A check valve; and
- Replacement (welding) of ISI class 2 check valve CK-ES3339, a High Pressure Safety Inspection pump minflow check valve.

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration (RVUHP) Inspection Activities

a. Inspection Scope

Since repairs were previously performed on control rod drive mechanism (CRDM) penetrations numbers 29 and 30, the Unit was considered to be in the high susceptibility ranking category and inspection activities were conducted in accordance with NRC Order EA-03-009. For each non-destructive examination (NDE) activity performed by the licensee with regard to the RVUHPs, the inspectors performed the following either through direct observation or through record review:

- Verified that the activities were performed in accordance with the requirements of NRC Order EA-03-009; and
- Verified that indications and defects, if detected, were dispositioned in accordance with the ASME Code or an NRC approved alternative (e.g., approved relief request).

In keeping with the Order, both visual examination (VT-2) and non-visual examinations eddy current (EC) and ultrasonic testing (UT)) were performed. The inspectors conducted a record review of the VT examination (procedures and still photographs), and the licensee’s criteria for confirming visual examination quality to ensure minimum examination coverage.

The inspectors also observed a sample of the non-visual examinations performed. In particular, the inspectors observed EC testing of the head vent, and UT testing of a minimum of 10 percent of the other 53 head penetrations, including a number of in-core instrumentation penetrations and CRDM penetrations, including those previously repaired. The inspectors reviewed the NDE examination procedures and confirmed that the calibration requirements (essential variables) were consistent with those used in vendor mockup demonstrations. The inspectors also reviewed the examination records to verify the extent of coverage of each penetration.

There were no examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service.

There were no welding repairs on the upper head penetrations completed since the beginning of the previous refueling outage.

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From April 1, 2006, through April 20, 2006, the inspectors reviewed the BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary."

The inspectors conducted a direct observation of BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. Specifically:

- On April 1, 2006, following shutdown, the resident inspector (RI) reviewed a sample of BACC visual examination activities through direct observation. This walkdown was completed with the Unit in Mode 3 at full operating pressure and included the lower containment building inner volume and annulus, and a review of the bottom vessel inspection records. The RI verified that the visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components; and

- The RI also reviewed the visual examination procedures and examination records for the BACC examination and verified that degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed the following boric acid leak corrective actions to confirm that they were consistent with the requirements of the ASME code and 10 CFR Part 50, Appendix B, Criterion XVI. The inspectors also reviewed the engineering evaluations performed for these same corrective action documents. The evaluations were verified, as applicable, to ensure that ASME Code wall thickness requirements were maintained:

- AR01021659, Component CV-1059, Pressurizer Spray Valve; and

- AR00891183, Component MO-3049, Safety Injection Tank T-82C Outlet Valve.

The documents reviewed during this inspection are listed in the attachment to this report. The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

From April 10, 2006, through April 18, 2006, the inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements.

The NRC inspectors observed acquisition of EC data, interviewed EC data acquisition personnel, and reviewed a sample of documents related to the SG ISI program to determine if:

- in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria were consistent with the Electric Power Research Institute (EPRI) 1007904, "Steam Generator In Situ Pressure Test Guidelines: Revision 2";

- the in-situ SG tube pressure testing screening criteria were properly applied in terms of SG tube selection based upon evaluation of the list of tubes with measured/sized flaws;

- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee's previous outage Operational Assessment predictions;

- the SG tube Eddy Current Examination (ET) examination scope and expansion criteria were sufficient to identify tube degradation based on-site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements and EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6";

- the SG tube ET examination scope included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, and support plates;

- the licensee identified new tube degradation mechanisms;

- the licensee implemented repair methods, which were consistent with the repair processes allowed in the plant TS requirements;

- the licensee primary-to-secondary leakage, (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;

- the licensee initiated evaluations for unretrievable loose parts;

- the EC probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination," of EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6; and

- the licensee identified deviations from EC data acquisition or analysis procedures.

The inspectors performed a review of SG ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the SG related problems;

- the licensee had established an appropriate threshold for identifying issues;

- the licensee had evaluated industry generic issues related to SG tube integrity; and

- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

Activities which were not applicable during this inspection and therefore not reviewed were:

- Inspection Procedure 71111.08, Step 02.04.d, associated with review of licensee activities for new SG tube degradation mechanisms. No new tube degradation mechanisms were identified; and

- Inspection Procedure 71111.08, Step 02.04.h, associated with review of corrective actions for primary-to-secondary leakage greater than 3 gallons per day. Primary-to-secondary leakage was below the minimum detectable threshold during the previous operating cycle.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

.1 Quarterly Review

a. Inspection Scope

The inspectors completed one inspection sample of licensed operator requalification training by observing a crew of licensed operators during simulator training on June 14, 2006. The inspectors assessed the operators' response to the simulated events which included a loss of main feedwater - manual control; evaluation of plant conditions for reactor shutdown when RPS setpoint not exceeded; loss of component cooling water; verification of containment isolation; and a faulted steam generator.

The inspectors verified that the operators were able to effectively mitigate the events through accurate and timely implementation of applicable alarm response procedures; Off-Normal Procedure 6.2, "Loss of Component Cooling;" Off-Normal Procedure 24.5, "Loss of Instrument AC Bus Y01;" Emergency Operating Procedure 1.0, "Standard Post Trip Actions;" and Emergency Operating Procedure 9.0, "Functional Recovery Procedure." The inspectors also observed the post-training critique to assess the licensee evaluators' and the crew's ability to self-identify performance deficiencies.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors completed one inspection sample pertaining to maintenance effectiveness by reviewing maintenance rule implementation activities for the following system and components:

- 2400 volt safety related switchgear

The inspectors reviewed the licensee's implementation of the maintenance rule requirements to verify that component and equipment failures were evaluated and appropriately dispositioned. The inspectors also verified that the selected systems and components were scoped into the maintenance rule and properly categorized as (a)(1) or (a)(2) in accordance with 10 CFR 50.65.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13Q)

a. Inspection Scope

The inspectors completed six inspection samples. The inspectors reviewed the following six activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee's risk assessment tool and risk categories in accordance with Administrative Procedure 4.02, Control of Equipment, Revision 29, and Fleet Procedure FP-OP-RSK-01, Risk Monitoring and Risk Management, Revision 0. Documents reviewed are listed in the attachment.

- Scheduled Orange Path due to PCS reduced inventory April 7, 2006
- Scheduled Orange Path due to PCS reduced inventory April 23, 2006
- Scheduled Yellow Path due to PCS vented on May 13, 2006
- Scheduled Yellow Risk due to EDG 1-2 surveillance test on May 24, 2006
- Elevated Risk due to planned power outage of Bus 96 (demineralized water transfer pump P-936 unavailable) on June 16, 2006
- Yellow Path due to EDG 1-2 inoperability (failure to meet start time) on June 22, 2006

The inspectors also verified that condition reports related to emergent equipment problems were entered into the corrective action program with the appropriate significance characterization. Select condition reports related to risk management during maintenance activities were reviewed to verify that planned corrective actions were appropriate and had been implemented as scheduled.

1R14 Operator Performance During Nonroutine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors completed four samples of operator performance during non-routine events. For the non-routine events described below, the inspectors attended pre-evolution briefs, reviewed operator logs and plant computer data, and observed operator performance to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures:

- Recovery of an incore instrument cask liner from the reactor cavity
- PCS Reduced inventory to midloop on April 7, 2006
- Operation at power and subsequent shutdown with a control rod fully inserted
- Troubleshooting of slow EDG start and subsequent restoration

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

For the three operability evaluations described in the Operability Recommendations (OPRs) listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the Updated Final Safety Analysis Report to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. In addition, the inspectors verified that the condition reports generated for equipment operability issues were entered into the licensee's corrective action program with the appropriate significance characterization. Documents reviewed are listed in the attachment.

- CCW Room environmental qualification issues OPR01021246-01
- Service Water piping downstream CV-0823 OPR01023662-00
- Missing clevis on reactor shielding

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the twelve post-maintenance tests listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). Further, the inspectors reviewed condition reports to verify that post maintenance testing problems were entered into the corrective action program with the appropriate significance characterization. For select condition reports, the inspectors verified that the corrective actions were appropriate and implemented as scheduled. Documents reviewed are listed in the attachment.

C Polar Crane Inspection following impact with Jib Crane
C Hot Leg Injection MOV, MO-3081
C CV-3070, HPSI Subcooling valve following replacement
C CV-826/823 Retest, SW flow balance
C Control Rod 26 Retest
C Control Rod 33 Relatch
C EDG 1-2 Retest after lube oil leak
C Control valve sump leakage sources outside containment
C P-7A Service Water Pump
C CV-0727, Auxiliary Feedwater flow control calibration
C MSIV solenoid replacement
C EDG 1-2 Retest after slow starting time

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Planned Refueling Outage

a. Inspection Scope

The inspectors observed and assessed licensee's performance in completing activities during a planned refueling outage that ended on May 10, 2006. This inspection constitutes one sample. The inspectors performed the following general activities periodically throughout the refueling outage:

- verified that plant equipment, including inventory control systems, required to minimize plant risk were aligned in accordance with plant procedures;

- performed walkdowns of the containment building to observe ongoing activities and verified closure capability;
- verified that equipment tagouts were adequate for activities that could affect shutdown risk;
- reviewed selected condition reports to verify that identified problems were accurately characterized; entered into the corrective action program with the appropriate significance; and that corrective actions were appropriate; and
- verified decay heat removal systems were aligned per TS.

The inspectors performed the following specific activities during the refueling outage:

- observed portions of refueling activities;
- randomly performed partial walkdowns of the spent fuel pool cooling system to verify it was properly aligned and not impacted by ongoing outage work;
- observed control room operator performance during portions of the primary coolant system drain down to mid-loop;
- while the plant was in mid-loop, the inspectors: (1) conducted plant walkdowns to verify to the extent practical that plant equipment required by GOP-14, "Shutdown Cooling Operations," Attachment 14, "Reduced Inventory Checklist," was available and properly aligned to minimize plant risk; (2) verified that containment closure capability was in place for the mitigation of radioactive releases, including appropriate staging of personnel and equipment; (3) verified that at least two independent, continuous indications of primary coolant system temperature and level were available; and (4) verified that at least two additional means of adding inventory to the primary coolant system were available, in addition to the residual heat removal system;
- periodically verified that station electrical power, emergency diesel generators, decay heat removal and primary coolant system inventory control systems were aligned as required by GOP-14;
- reviewed documentation to verify that appropriate mode change checklists were appropriately completed during plant startup;
- performed containment walkdowns prior to plant restart to evaluate the licensee's process for maintaining containment cleanliness as required by Checklist 1.4, "Containment Closeout Walk-Through," and to verify that no material was left in containment that could adversely impact containment sump design attributes; and
- observed operator performance during portions of reactor startup, turbine valve testing, and main generator synchronization to the electrical grid.

b. <u>Findings</u>

<u>Introduction</u>

A self-revealed violation of 10 CFR 50 Appendix B, Criterion XVI, of very low safety significance (Green) was identified on April 15, 2006, when licensee personnel noted failed fuel cladding in a location that had previously experienced a fuel clad failure. A failure had occurred in the same location in 1993 that resulted in loss of fuel pellets. A corrective action for this significant condition adverse to quality had been removed, thus allowing recurrence.

Description

On April 15, 2006, the licensee noted a fuel cladding breach on one fuel rod, on fuel assembly (T30), at location B19 adjacent to the core support barrel. The cladding breach occurred on the upper end of the fuel rod and exposed the fuel plenum spring. No fuel pellets were dislodged from the fuel rod and the only other damage occurred to spacer assemblies. The licensee inspected the rest of the assemblies and no other assemblies or fuel rods were impacted. The fuel assembly was offloaded, per the licensee's reload plan, and placed in the spent fuel pool.

In 1993, damage occurred to the fuel rod (S15) in the same location of B19. This event resulted it the loss of approximately 215 pellets and fragmentation of the rod. As a result of the damaged fuel rod in 1993, the NRC sent an Augmented Inspection Team to investigate the event (IR 50-255/93018). The licensee classified the issue as a Significant Condition Adverse to Quality (CAP 010123) and conducted a root cause, but could not determine an exact root cause. The licensee determined the most likely cause was some interaction between the fuel and the core shroud based on interferences between the shroud, the fuel assembly or the upper guide structure. In order to protect fuel from the interaction, the licensee placed steel pins in several locations where the interaction was most likely. Over the course of several cycles, the licensee reduced the number of steel rods, eventually ending the use of the steel rods in Cycle 15 (three cycles ago). The licensee based this decision on the lack of damage to the steel rods.

The licencee wrote AR01024290 (April 15, 2006) to address the newly identified damage to the fuel rod and the changes to the corrective actions taken subsequent to the 1993 issue. The licensee believes the two events are related to interference problems between the fuel assembly and the core support barrel. The immediate corrective action included installing steel pins in the susceptible location during core reload. The licensee added steel pins in one other susceptible location. A formal root cause will evaluate the condition.

Analysis

The inspectors concluded that the removal of the corrective action to maintain steel rods in certain fuel locations was a performance deficiency since it allowed recurrence of a significant condition adverse to quality. The inspectors concluded that the issue was more than minor since the clad on one fuel rod was breached. This condition directly affects the barrier cornerstone objective of fuel cladding integrity. In accordance with IMC 0609, Significance Determination Process, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-power Situations," Attachment 1, the inspectors concluded the finding was of very low safety significance since the finding only affected the fuel barrier and was bounded by current analysis.

Enforcement

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires that measures be established to assure that conditions adverse quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Further, for significant conditions adverse to quality (SCAQ), measures shall assure the cause of the condition is determined and action taken to prevent repetition. Contrary to this requirement, the licensee removed corrective action for an SCAQ thus allowing recurrence. Specifically, in 1993, a significant condition adverse to quality occurred when fuel clad failed and specific corrective actions were put in place to prevent recurrence. Subsequently, the licensee removed the corrective action of installation of steel rods at certain fuel locations, thus establishing conditions for the fuel clad to fail again in 2006. Therefore, the inspectors determined that this finding was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because this violation was of very low safety significance (Green) and was documented in the licensee's corrective action program as Condition Reports CAP AR01024290, this finding is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000255/2006004-02)

Introduction

On May 11, 2006, following restart after a refueling outage, the licensee identified an abnormal core flux distribution and decided to shut down the plant. Subsequent investigation revealed that rod 33 had not been latched following head installation.

Discussion

On May 11, 2006, while at 24 percent power, the licensee calculated that quadrant power tilt, a measure of core flux distribution, was at 5 percent. After reviewing additional data, the licensee suspected that rod 33 was unlatched. In order to confirm this suspicion, the licensee shut down the reactor. While shut down, the licensee validated that rod 33 had not been properly latched. After latching the rod and confirming proper latching through weight measurements, the licensee restarted the reactor. Subsequent evaluation of the computer program to monitor core flux indicated that the computer program suppressed input from several in core detector locations. Once included in the analysis, quadrant power tilt was 8 percent.

During the forced outage, the licensee reviewed the rod latching procedure and concluded that due to the nature of the process to latch a rod, several error traps existed that could lead to a failure to latch a rod. As part of the root cause investigation, the licensee reviewed videotape of the control rod latching process. This review identified several errors in the performance of the work. Most notably, although the workers performed the latch verification step, the workers did not recognize that the ability to rotate the tool indicated the rod was not latched.

Analysis

The inspectors concluded that reactor power operations with a rod not properly latched represented a performance deficiency that warranted a safety significance determination. The inspectors concluded that the issue was more than minor because the fully inserted rod resulted in an abnormal flux distribution in the core. This flux distribution affects several LCOs, including quadrant power tilt and total radial peaking factor. Due, in part, to power remaining less than 25 percent, these TSs were not violated. The inspectors reviewed the finding in accordance with IMC 0609 to determine the safety significance. Although the finding affected both the barrier integrity and initiating event cornerstones, the inspectors concluded that because the power level remained below 25 percent and no thermal limits were exceeded, the finding was of very low safety significance. This finding included a cross-cutting aspect in the area of human performance in that work practices used were inadequate to discover the rod was not latched.

Enforcement

Technical Specification 5.4.1., required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33. Appendix A, Item 9.a., states that maintenance that can affect the performance of safety-related equipment should be preplanned and performed in accordance with written procedures appropriate to the circumstances. This includes procedures such as RFL-R-11, Coupling Control Rods, which provides instructions on coupling of control rods. Contrary to the requirements of this procedure, the workers failed to recognize the coupling check of the control rod indicated the rod had not been coupled. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (AR 01029611), this violation is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000255/2006004-03). The licensee's initial corrective action included shutting down the reactor and recoupling the affected rod.

.2 Forced Outage

a. On May 11, 2006, the licensee entered a forced outage to latch a control rod that had not been latched during the previous refueling outage. The inspectors observed control room activities during shutdown. The inspectors also completed a walkdown of accessible portions of containment with site personnel. The inspectors evaluated these activities to ensure licensee personnel was performing within TS requirements, plant procedures, and other applicable requirements. The inspectors observed cooldown and heatup activities to verify the licensee conducted these activities consistent with TS requirements. The inspectors observed control room activities during restart. The inspectors also completed a closeout walkdown of accessible portions of containment with site personnel. The inspectors evaluated these activities to ensure licensee personnel were performing within TS requirements, plant procedures, and other applicable requirements.

This represented one sample; coupled with the refueling outage, two samples of this inspection procedure were completed.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed five surveillance tests and/or reviewed test data of selected risk-significant structures, systems, and components (SSCs), listed below, to assess, as appropriate, whether the SSCs met the requirements of the TS; the USAR; Palisades Administrative Procedure 9.20; Technical Specification Surveillance and Special Testing Program; Engineering Manual EM-09-02 and EM-09-04, Inservice Testing of Plant Valves and Inservice Testing of Selected Safety Related Pumps. One of the samples was an in-service test (IST). One of the samples was a containment isolation valve. The inspectors also determined whether the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. Further, the inspectors reviewed selected condition reports regarding surveillance testing activities. The inspectors verified that the identified problems were entered into the licensee's corrective action program with the appropriate significance characterization and that the planned and completed corrective actions were appropriate. Additional documents reviewed are listed in the attachment.

- RT-8D, Engineered Safeguards System- Right Channel
- RO-105, Full Flow Test for SIT Check Valves and PCS Loop Check Valves (IST)
- RT-129, Functional Test of Bus 1C Undervoltage Relays
- RO-32-47, LLRT - Local Leak Rate Test for Penetration MZ-47 (containment isolation valve)
- RT-92, Containment Sump Inspection and Close Out

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors completed one baseline inspection sample by reviewing the following temporary modification:

- Temporary Modification EC 544, "Temporary Jumper of Test Switch Contacts on TPS-187Z-107"

The inspectors reviewed the design documents and 10 CFR 50.59 safety screening to verify that the temporary modification did not affect the operability of the related systems and other interfacing systems. The inspectors reviewed documentation to verify that the modification was implemented as designed. Post modification testing results were reviewed to verify that the system functioned as intended after the modification was implemented.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified four radiologically significant work areas within radiation areas, high radiation areas (HRA) and airborne areas in the auxiliary and containment buildings. Selected work packages and radiation work permits (RWP) were reviewed to determine if radiological controls including surveys, postings, air sampling data, and barricades were acceptable. Work areas included:

- Work Order 26447; RX Head Insp - Under Reactor Head Entries;
- Work Order 26964; E-50B, Perform Eddy Current Testing/Tube Plugging;
- Work Order 26963; E-50A, Perform Eddy Current Testing/Tube Plugging; and
- Work Order 25806; Removal of Incores and RVLMS Probes.

This review represented one sample.

The identified radiologically significant work areas were walked down and surveyed to determine if the prescribed RWP, procedures, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located. This review represented one sample.

The inspectors reviewed selected RWPs, and associated radiological controls used to access these and other radiologically significant areas, and evaluated the work control instructions and control barriers that were specified in order to determine if the controls and requirements provided adequate worker protection. Site TS requirements for HRAs and locked high radiation areas were used as standards for the necessary barriers. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors attended pre-job briefings to determine if instructions to workers emphasized the actions required when their electronic dosimeters noticeably malfunctioned or alarmed. This review represented one sample.

The inspectors reviewed job planning records and interviewed licensee representatives to determine if there were airborne radioactivity areas in the plant with a potential for individual worker internal exposures of >50 millirem committed effective dose equivalent. Barrier integrity and engineering controls performance, such as high efficiency particulate filtration ventilation system operation and use of respiratory protection, were evaluated for worker protection. Work areas having a history of, or the potential for, airborne transuranic isotopes were reviewed to determine if the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection. This review represented one sample.

The adequacy of the licensee's internal dose assessment process for internal exposures >50 millirem committed effective dose equivalent was assessed to determine if affected personnel were properly monitored utilizing calibrated equipment and that the data was analyzed and internal exposures were properly assessed in accordance with licensee procedures. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and condition reports related to the access control program to determined if identified problems were entered into the corrective action program for resolution. This review represented one sample.

Corrective action reports related to access controls and high radiation area radiological incidents (non-performance indicator occurrences identified by the licensee in HRAs <1Rem/hr) were reviewed. Staff members were interviewed and corrective action documents were reviewed to determine if follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of Non-Cited Violations tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

This review represented one sample.

The inspectors evaluated the licensee's process for problem identification, characterization, prioritization, and determined if problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies identified in the problem identification and resolution process, the inspectors determined if the licensee's self-assessment activities also identified and addressed these deficiencies. This review represented one sample.

The inspectors discussed performance indicators with the radiation protection staff and reviewed data from the licensee's corrective action program to determine if there were any performance indicators for the occupational exposure cornerstone that had not been reviewed. There were none to evaluate. The licensee is evaluating an event that occurred on April 18, 2006, as a potential performance indicator occurrence, as discussed in Section 4OA7 of this report. This evaluation will be completed before submitting the performance indicator data to the NRC for the second quarter 2006. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

The inspectors evaluated selected jobs being performed in radiation areas, potential airborne radioactivity areas, and HRAs for observation of work activities that presented the greatest radiological risk to workers and included areas where radiological gradients were present (Section 2OS1.1). This involved work that was estimated to result in higher collective doses, and included vessel head decontamination, steam generator inspection, and incore instrumentation activities, other selected work areas in the containment building.

The inspectors reviewed radiological job requirements including RWP and work procedure requirements, and attended as-low-as-is-reasonably-achievable (ALARA) job briefings. Job performance was observed with respect to these requirements to determine if radiological conditions in the work areas were adequately communicated to workers through pre-job briefings, and radiological condition postings. This review represented one sample.

The inspectors also evaluated the adequacy of radiological controls including required radiation, contamination and airborne surveys for system breaches, and entry into HRAs. Radiation protection job coverage, including direct visual surveillance by radiation protection technicians, along with the remote monitoring and teledosimetry systems and contamination control processes, were evaluated to determine if workers were adequately protected from radiological exposure. This review represented one sample.

Work in high radiation areas having significant dose rate gradients was observed to evaluate the application of dosimetry to effectively monitor exposure to personnel, and to determine if licensee controls were adequate. The inspectors observed radiation protection coverage of the vessel head decontamination and steam generator work, which involved controlling worker locations based on radiation survey data and real time monitoring using teledosimetry in order to maintain personnel radiological exposure ALARA. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 <u>High Risk Significant, High Dose Rate High Radiation Area, and Very High Radiation Area Controls</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's performance indicators for high risk, high dose rate HRAs, and for very high radiation areas to determine if workers were adequately protected from radiological over-exposure. Discussions were held with radiation protection management concerning high dose rate HRA, including procedural changes that had occurred since the last inspection. This was done to determine if any procedure modifications had substantially reduced the effectiveness and level of worker protection. This review represented one sample.

During plant walkdowns, the posting and locking of entrances to high dose rate HRA and very high radiation areas were reviewed for adequacy. This review represented one sample.

b. <u>Findings</u>

No findings of significance were identified.

.5 <u>Radiation Worker Performance</u>

a. <u>Inspection Scope</u>

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements. The inspectors also evaluated whether workers were aware of the significant radiological conditions in their workplace, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present. This review represented one sample.

Radiological problem reports, which found that the cause of an event resulted from radiation worker errors, were reviewed to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. This review represented one sample.

b. <u>Findings</u>

No findings of significance were identified.

.6 <u>Radiation Protection Technician Proficiency</u>

a. <u>Inspection Scope</u>

The inspectors observed and evaluated RP technician performance with respect to RP work requirements. This was done to evaluate whether the technicians were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities. This review represented one sample.

Radiological problem reports, which found that the cause of an event was RP technician error, were reviewed to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. This review represented one sample.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends along with ongoing and planned activities in order to assess current performance and exposure challenges. This included determining the plant's current 3-year rolling average collective exposure and comparing the site's radiological exposure on a yearly basis for the previous 5 years. This review represented one sample.

The inspectors reviewed the outage work scheduled during the inspection period along with associated work activity exposure estimates including four work activities which were likely to result in the highest personnel collective exposures. This review represented one sample.

Site specific trends in collective exposures and source-term measurements were reviewed. This review represented one sample.

Procedures associated with maintaining occupational exposures ALARA, and processes used to estimate and track work activity specific exposures were reviewed. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities, ranked by estimated exposure, that were in progress and selected the four work activities of highest exposure significance. This review represented one sample.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to determine if the licensee had established procedures, along with engineering and work controls, that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, or special circumstances. This review represented one sample.

The interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups were evaluated to identify interface problems or missing program elements. This review represented one sample.

The integration of ALARA requirements into work procedures and RWP documents was evaluated to determine if the licensee's radiological job planning would reduce dose. This review represented one sample.

Shielding requests from the radiation protection group were evaluated with respect to dose rate reduction and reduced worker exposure, along with engineering shielding responses follow up. This review represented one sample.

The inspectors reviewed work activity planning to determine if there was consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components and piping, job scheduling, along with shielding and scaffolding installation and removal activities. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors selected four work activities in radiation areas, potential airborne radioactivity areas, and HRAs for observation, emphasizing work activities that presented the greatest radiological risk to workers. Jobs that were expected to result in significant collective doses and involved potentially changing or deteriorating radiological conditions were observed. These included vessel head decontamination, steam generator inspection, and in core instrumentation activities. The licensee's use of ALARA controls for these work activities was evaluated using the following:

- The use of engineering controls to achieve dose reductions was evaluated to determine if procedures and controls were consistent with the ALARA reviews; that sufficient shielding of radiation sources was provided for, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding. This review represented one sample.

- Job sites were observed to determine if workers were utilizing the low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays. This review represented one sample.

- The inspectors attended ALARA pre-job briefings and observed ongoing work activities to determine if workers received appropriate on-the-job supervision to ensure the ALARA requirements were met. This included determining if the first-line job supervisor ensured that the work activity was conducted in a dose efficient manner by minimizing work crew size, ensuring that workers were properly trained, and that proper tools and equipment were available when the job started. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and determined if the licensee was making allowances and had developed contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry. This review represented one sample.

The inspectors determined if the licensee had developed an understanding of the plant source-term, which included knowledge of input mechanisms in order to reduce the source term. The licensee's source-term control strategy, which included cobalt reduction plus a shutdown and operating chemistry plan which can minimize the source term external to the core, was evaluated. Other methods used by the licensee to control the source term, including component/system decontamination, hot spot flushing and the use of shielding, were evaluated. These reviews represented one sample.

The licensee's process for identification of specific sources was reviewed along with exposure reduction actions and the priorities the licensee had established for implementation of those actions. Results achieved against these priorities since the last refueling cycle were reviewed. For the current assessment period, source-term reduction evaluations were verified, and actions taken to reduce the overall source-term were compared to the previous year. These reviews represented one sample.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance

a. Inspection Scope

Radiation worker and RP technician performance was observed during work activities being performed in radiation areas, airborne radioactivity areas, and HRAs that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas, and that work activity controls were being complied with. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved. This review represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in other sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that condition reports were being generated and entered into the corrective action program with the appropriate significance characterization. For select condition reports, the inspectors also verified that identified corrective actions were appropriate and had been implemented or were scheduled to be implemented in a timely manner commensurate with the significance of the identified problem.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP action requests to identify trends that could indicate the existence of a more significant safety issue. The inspectors also reviewed the Operations Department Monthly Performance Report dated March 2006, the Site Department Roll-up Meeting (DRUM) Report for the 1st Quarter of 2006, and the Corrective Action Program Performance Indicator Summary dated May 2006. The inspectors' review for potential trends included the results from the daily inspector CAP item screening discussed in Section 4OA2.1. The plant CAP action request screening meetings were observed to review the licensee's level of effort in identifying potential trends, and any associated corrective actions that were planned or implemented. In addition, the inspectors reviewed issues documented outside the normal CAP that included maintenance work orders, component status reports, performance indicators, and Operations control room logs. The inspectors' review nominally considered the 6 month period of January through June 2006. The inspectors compared and contrasted their results with the results obtained by the licensee during previous internal reviews.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 On April 19, 2006, the inspectors observed licensee response to an incident involving handling of cask containing used incore instrumentation. While submerged in the reactor cavity in preparation for adding additional used incore instruments, the liner, propelled by buoyant forces, rose from the cask and briefly surfaced. After several seconds, the liner sank to the floor of the reactor cavity. Due to the high radiation levels of the cask, workers handling the cask evacuated the area and the licensee evacuated containment. The licensee reported this event to the NRC as Event Number 42514 on April 19, 2006. The inspectors reviewed the licensee response to the event, including evaluation for EAL entry and immediate actions to protect workers and the public. The NRC performed a special inspection on this event which is documented in NRC Inspection Report 0500255/2006008.

.2 On April 22, 2006, the inspectors observed licensee response and recovery actions related to an impact between the polar crane bridge and the boom of the containment jib crane. As a result of the impact, both cranes were stopped with suspended loads. The inspectors observed licensee's actions to safely lower the loads and to separate the cranes. The inspectors evaluated the licensee's actions to protect workers and the public.

Introduction

On April 22, 2006, while moving equipment in containment, the polar crane bridge struck and severely damaged the jib crane. Loads attached to the jib crane were lifted and minor damage occurred to the upper structure of the reactor head. The licensee disengaged the two cranes and lowered the attached loads without further damage.

Discussion

On April 22, 2006, while shut down with the reactor cavity drained, and the reactor vessel head on the inspection stand, the licensee was removing equipment from the reactor cavity using the polar crane. At the same time, the licensee was preparing to move a jacking beam from the vessel head using the jib crane. The reactor vessel head was not being lifted during this time. In this configuration, the jib crane was fully extended and in the plane of movement of the polar crane. Due to poor communication between the crane operators, the polar crane operator rotated the bridge of the polar into the jib crane. The force of the polar crane distorted the boom of the jib crane and caused the attached load to rise several feet. A worker descending from the jack beam on a ladder jumped to the ground to avoid injury as the jack beam was raised. The crane cables were dragged across portions of the upper head structure causing minor damage to a empty cable tray and a hand rail at the top of the structure. The licensee successfully lowered the loads attached to each of the cranes and disengaged the cranes.

After the cranes were disengaged, the licensee inspected both cranes as well as the vessel head. The polar crane suffered only minor damage; however, significant damage was done to the jib crane and it had to be removed from containment for repair. The licensee repaired the damage to the upper portions of the vessel head. However, the licensee failed to identify that bolts associated with the jacking crane had fallen onto the head, and when the head was later moved to the vessel, one of the bolts fell into the reactor cavity and was retrieved by the licensee.

Analysis

The inspectors concluded that the licensee’s failure to control crane operations resulted in damage to the facility and warranted a safety significance determination. The inspectors concluded that the issue was more than minor because the licensee’s failure to control crane motion could reasonably be seen as a precursor to a more significant event. Specifically, failure to understand travel paths for all crane parts increases the likelihood of a heavy load drop. In addition, this event resulted in damage to plant equipment, including a jib crane and minor damage to the upper structure of the vessel head. Finally, the event resulted foreign material exclusion (FME) entry into the reactor cavity and could have resulted in loss of the FME into the reactor vessel. The inspectors reviewed the finding in accordance with IMC 0609, Appendix G. The inspectors concluded that the finding did not satisfy any of the criteria that would require a Phase 2 or 3 analysis. The inspectors noted that no heavy load lift was in progress and no impact occurred to safety-related systems. Therefore, the inspectors in consultation with NRC management, concluded the finding was of very low safety significance. This finding impacted the cross-cutting area of human performance because the licensee failed to use sufficient human performance tools to prevent the cranes from coming into contact.

Enforcement

Technical Specification 5.4.1.a required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33. Appendix A, Item 9.a., states that maintenance that can affect the performance of safety-related equipment should be preplanned and performed in accordance with written procedures appropriate to the circumstances. Contrary to this requirement, the licensee procedures for crane operation in containment failed to address movement of loads that did not meet the requirements of heavy loads. The licensee extended the boom of the crane into the plane of the polar crane bridge. However, because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (AR 01025665) this violation is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000255/2006004-04). The licensee's initial corrective action included disengaging the cranes and inspecting and repairing the vessel head.

4OA5 Other Activities

1. (Closed) URI 05000255/2003002-04, "Nonconformance with Systematic Evaluation Program Commitment"

a. Inspection Scope

The inspectors reviewed applicable documents related to this URI, including the license basis for Palisades. After discussion with the Office of Nuclear Reactor Regulation, the inspectors concluded that the licensee had improperly implemented 10 CFR 50.59 when the requirement pump for P-5 operability was removed from the FSAR.

b. Findings

Introduction

In February of 2003, the NRC identified an issue regarding the licensee's compliance with commitments for maintenance of pump P-5, a warm water recirculation pump. Because of apparent discrepancies with meeting the commitments, the NRC opened URI 05000255/2003002-04. After further review, the inspectors have concluded that the licensee failed to use the 10 CFR 50.59 process when the P-5 pump was removed from the FSAR as a required component; therefore, a Severity Level IV NCV was warranted. This item is closed.

Description

In NRC Inspection Report 0500255/2003002, the inspectors documented URI 2003-02-04 regarding the inability of pump P-5 to take suction from the lake. The NRC had accepted the licensee commitment to maintain this capability during the Systematic Evaluation Program (SEP) (NUREG 0820) in lieu of inspections of the intake structure as identified in Regulatory Guide 1.127. In NUREG 0820, the NRC concluded that the licensee's proposal to maintain and test the capabilities of P-5 was equivalent to applicable inspections stipulated in Regulatory Guide 1.127. Based on this conclusion, the NRC determined that this element of the SEP was safe. In 2000, the licensee revised the FSAR to delete the requirement to maintain the P-5 pump. The deletion of the P-5 pump requirements without addition of the Regulatory Guide 1.127 inspections resulted in a change to the facility that required prior NRC approval. In 2003, the licensee revised the FSAR to include annual intake structure inspections following the annual ice floes, thus re-establishing some of the applicable inspections. Specifically, the licensee did not recognize that special inspections were required following unusual events that could damage the intake structure. The licensee has subsequently revised the FSAR to include the applicable inspections in Regulatory

Guide 1.127. Thus far, the inspections performed have identified some damage to the intake structures but the damage has not impacted operability of the intake structure.

The inspectors reviewed the inspection history of the intake structure from 2000 until 2006. Between December 2002 and March 2003, the P-5 pump could not take suction from the lake nor was there a current inspection for the intake structure. The licensee performed an intake structure inspection in the spring of 2003 that confirmed the operability of the structure. Further, no unusual events occurred, aside from the annual ice floes, that would have triggered additional inspections of the intake structure.

Analysis

The inspectors concluded that the failure to perform an adequate 50.59 review on the removal of the P-5 pump from the FSAR was a performance deficiency that warranted a significance review. The inspectors concluded that the SEP conclusion of overall plant safety relied upon the maintenance of the P-5 pump in lieu of inspection identified in Regulatory Guide 1.127. The removal of P-5 from FSAR without addition of a countervailing capability represented a change to the facility that required prior NRC approval. Since subsequent inspections of the intake structure consistent with Regulatory Guide 1.127 requirements indicated the structure remained operable, no loss of function occurred. Therefore, the inspectors concluded this issue impacted the NRC's ability to perform its regulatory function. This item has some potential to impact the mitigating system cornerstone; it was assessed using IMC 0609 to determine the impact if any on mitigating systems. Since the pump is not relied upon during plant licensing basis events, and the intake structure is being inspected to ensure its safety function is maintained, there is no significant impact to the objective to ensure availability of the ultimate heat sink. Consistent with the Enforcement Policy, the inspectors concluded that this finding was of very low safety significance (Green).

Enforcement

10 CFR 50.59 states, in part, that the licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change test or experiment if the change, test, or experiment would: ...(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated). Contrary to this requirement, the licensee deleted the requirement to maintain P-5 pump available. As part of the NRC's conclusion of the acceptability of site safety documented on NUREG-0820, the NRC relied upon maintenance of the availability of the P-5 pump in lieu of inspections stipulated in Regulatory Guide 1.127. After recognition that this change required prior NRC approval, in 2003 the licensee added a subset of the inspections stipulated in Regulatory Guide 1.127 to the FSAR.

In 2006, the licensee added the remainder of the inspections stipulated in Regulatory Guide 1.127 and referenced in NUREG 0820. The inspectors concluded that since the intake structure remained operable, the violation was of very low safety significance and classified the violation of 10 CFR 50.59 as a Severity Level IV violation. This item has been entered into the licensee's corrective action program (AR 01032931). This violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC

Enforcement Policy (NCV 05000255/2006004-05). The licensee has inspected the intake structure and revised the FSAR.

2. Failed EDG Snubber

The inspectors identified an unresolved item associated with the monthly diesel surveillance test conducted on November 20, 2005. A fuel leak developed on the 1-2 EDG that resulted in a failure of the surveillance test. The cause was related to a part which had been installed 28 days earlier. The part has been replaced, and there are no other susceptible parts in the diesel engines on site. The licensee wrote Licensee Event Report 2005-007-00 on this issue. This is an unresolved issue pending assessment of the licensee's evaluation by the NRC.

Description

On November 20, 2005, while conducting a monthly surveillance test of the 1-2 EDG, a fuel leak developed on the fuel injector for the 5R cylinder. Licensee personnel determined the fuel leak was significant, and a potential fire hazard, and secured the EDG. Troubleshooting identified that the snubber valve for 5R fuel pump, which had been replaced 28 days earlier, had cracked and caused the fuel oil spray. The licensee replaced the failed snubber and verified operability of the EDG. Subsequent testing of the snubber indicated that an improper heat treatment had been applied to the snubber.

The licensee contacted the EDG vendor that had supplied the snubber. The vendor reviewed the data related to the valve and performed additional testing. Based on the result, the vendor confirmed that the valve had been properly installed and the improper heat treatment had caused the valve's failure. However, the vendor could not correlate the obtained data to any snubber they had produced since 1995. The vendor hypothesized that the licensee's stock may have been contaminated with stock from a batch produced in the 1990's. Previous OE in 1993 from Diablo Canyon had identified bad snubbers. The licensee completed their assessment in late June 2006 which concluded the snubber failure did not render the EDG inoperable. The inspectors will review the licensee's assessment. This issue will be treated as URI 05000255/2006004-06 pending completion of the inspectors' evaluation of the assessment.

.3 Failed High Pressure Safety Injection (HPSI)

The inspectors identified an URI concerning the adequacy of a design implemented to ensure functionality of CV-3070, the left train HPSI sub-cooling control valve. Historical records reveal that the air operator for CV-3070 had previously been supported by a spring hanger that was removed in 1994. The valve failed to stroke on March 29, 2006. The licensee's root cause analysis determined the failure mechanism for the valve not stroking was failure to adequately support the valve.

Description

On March 29, 2006, control valve CV-3070, left train HPSI sub-cooling valve for HPSI pump P-66B, failed to open during preventive maintenance. Maintenance was being performed to inspect and repair the valve's oiler. Maintenance visually inspected the valve's oiler and requested Operations to test stroke the valve. The test stroke was attempted, and the valve failed to open. As a result of CV-3070's failure to open, the plant entered the applicable 72-hour limiting condition for operation action 3.5.2B.1, that requires the train to be restored to operable. Failure to complete the LCO action required the plant to shut down. This issue was reported to the NRC as Event Number 42462.

Control valve CV-3070 would not open until it was mechanically agitated. The subsequent diagnostic testing of CV-3070 showed the stem movement of the valve was not smooth indicating internal valve component interference. The valve is mounted in a vertical pipe section with its air operator oriented horizontally. The lack of adequate support of the valve operator led to sagging of the valve operator and the bottom of the stem interacting with the backseat surface.

The hanger supporting CV-3070 presumably was installed to support the air operator. A search of historical records at the time found no documentation showing the hanger existed. The hanger was later removed on the implementation of specification change SC-93-068 in October 1994. In 2003 the valve also failed to stroke. Valve diagnostics had been showing some degradation of the valve. Subsequent to the March 2006, failure of CV-3070, a determination was made to replace the valve body, stem, and valve wedges. Additionally, a new spring hanger support was designed. Replacement of the valve and installation of the new spring hanger was completed during the 2006 refueling outage. Diagnostic testing following the valve replacement and hanger installation has resulted in proper valve operation. Since the valve is performing its safety, there is no current issue. The licensee's analysis to determine if CV-3070 was functional in its past design configuration had not been completed by the end of the inspection period. Therefore, this issue is considered an (URI 05000255/2006004-07) pending completion of this analysis and the NRC's review of the analysis.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. P. Harden and other members of licensee management on June 10, 2006. Licensee personnel acknowledged the findings presented. The inspectors asked licensee personnel whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Access control to radiologically significant areas, and the ALARA planning and controls program with Mr. P. Harden on April 14, 2006.
- Baseline procedure IP 71111.08 with Mr. P. Harden and other members of licensee management on April 21, 2006. The inspectors returned proprietary

information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Occupational Radiation Safety

Technical Specification 5.7.2 requires that high radiation areas greater than 1.0 rem/ hour at 30 centimeters from the radiation source and within a Regulatory Guide area where no enclosure can be reasonably constructed shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated as a warning device. Technical Specification 5.4.1 requires that written procedures be established and implemented for activities provided in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include radiation protection procedures for access control to radiological areas, which are provided by licensee procedure Administrative Procedure 7.13, "Radiological Area Access"

Revision 11. The procedure requires that workers ensure and verify that swing gates are returned to the closed position after passing through. Contrary to these requirements, on April 18, 2006, two individuals working in a high radiation area rotated a swing gate to move a required piece of equipment and failed to return the swing gate to the original configuration. This condition defeated the barricade and the area was not conspicuously posted. This incident is documented in the licensee’s corrective action program as AR 01024675. This issue represents a finding of very low safety significance because it did not involve ALARA planning or work controls, there was no overexposure or substantial potential for an overexposure to the workers that entered the high radiation area, nor was the licensee’s ability to assess worker dose compromised.

Technical Specification 5.4.1 requires that written procedures be established and implemented for activities provided in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include radiation protection procedures for personnel monitoring, which are provided by licensee procedure FP-RP-DP-01, "Dosimetry Program," Revision 0. The procedure requires that workers wear the assigned thermoluminescent dosimeters (TLD) at all times when in the Radiologically Controlled Area (RCA). Contrary to these requirements, on March 20, 2006, an individual conducted radiography activities inside the RCA without wearing the assigned TLD. This incident is documented in the licensee’s corrective action program as AR 01019453. This issue represents a finding of very low safety significance because it did not involve ALARA planning or work controls, there was no overexposure or substantial potential for an overexposure to the worker, nor was the licensee’s ability to assess worker dose compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
P. Harden, Site Vice President
M. Acker, Engineering Programs (ISI)
T. Blake, Nuclear Safety Assurance Manager
M. Carlson, Engineering Director
B. Dotson, Regulatory Compliance
G. Baustian, Training Manager
W. Godes, Operations Fleet Simulator Supervisor
J. Hager, Engineering Programs (SG)
G. Hettel, Plant Manager
L. Lahti, Licensing Manager
G. Lofthus, Fleet NDE
D. Malone, Regulatory Affairs
C. Moeller, Radiation Protection General Supervisor
B. Patrick, Radiation Protection Manager
P. Schmidt, Simulator Training Supervisor
G. Sturm, ALARA Specialist
J. Walker, Licensed Operator Requalification Training Supervisor
K. Yeager, Assistant Operations Manager

Nuclear Regulatory Commission
M. Padovan, Project Manager, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000255/2006004-01	URI	Isolation of one Shutdown Cooling Heat Exchanger (Section 1R04)
05000255/2006004-02	NCV	Failure to Comply with 10 CFR 50 Appendix B, Criterion XVI for Recurring Fuel Assembly Cladding Failure (Section 1R20)
05000255/2006004-03	NCV	Power Operations with One Rod Unlatched Due to an Inadequate Procedure (Section 1R20)
05000255/2006004-04	NCV	Polar Crane Struck Jib Crane (Section 4OA3)
05000255/2006004-05	NCV	Failure to Comply with 10 CFR 50.59 for P-5 Removal from the FSAR (Section 4OA5.1)

05000255/2006004-06	URI	Failure of Component on 1-2 EDG Causes Surveillance Failure (Section 4OA5.2)
05000255/2006004-07	URI	Control Valve CV-3070 Fails to Stroke (Section 4OA5.3)

Closed

05000255/2006004-02	NCV	Failure to Comply with 10 CFR 50 Appendix B, Criterion XVI for Recurring Fuel Assembly Cladding Failure (Section 1R20)
05000255/2006004-03	NCV	Power Operations with One Rod Unlatched Due to Violation of TS 5.4, Inadequate Procedure (Section 1R20)
05000255/2006004-04	NCV	Polar Crane Struck Jib Crane (Section 4OA3)
05000255/2006004-05	NCV	Failure to Comply with 10 CFR 50.59 for P-5 Pump Removal from the FSAR (Section 4OA5.1)
05000255/2003002-04	URI	Nonconformance with Systematic Evaluation Program Commitment (Section 4OA5.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a documents on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection
SOP-23, Attachment 10, Warm Weather Checklist, Revision 22, May 28, 2006
Palisades Site Summer Readiness Report, June 26, 2006

1R04 Equipment Alignment
Palisades Nuclear Administrative Procedure 4.02, Control of Equipment, Revision 29
SOP-3, Safety Injection and Shutdown Cooling System, Revision 66
Clearance Order 00013993, Work On CV-3070, Revision 0
EC7866, Enhancement to Technical Specification Basis on Shutdown Cooling Train Definition, Revision 0
Palisades System Operating Procedure No SOP-15, Service Water System, Revision 37
Palisades System Operating Procedure No SOP-16, Component Cooling Water System, Revision 28
Piping and Instrument Diagram —208, Service Water System, Sheet 1A, Revision 55
Piping and Instrument Diagram —209, Component Cooling Water System, Sheet A, Revision 7
SOP-24, Ventilation and Air Conditioning System, Revision 44

1R05 Fire Protection
Palisades Nuclear Plant Fire Hazards Analysis, Revision 5

1R08 Inservice Inspection Activities
NDT-VT-01, Visual Examination, Revision 15
NDT-PT-01, Liquid Penetrant Examination, Revision 15
NDT-MT-01, Magnetic Particle Examination, Revision 13
UT-02, Ultrasonic Examination of Vessel Welds and Adjacent Base Metal, Revision 26
NDT-UT-32, Ultrasonic Examination of Ferritic Pipe Welds, Revision 3
EM-09-14, Palisades Nuclear Plant Engineering Manual Procedure, Revision 5
RT-71A, Primary Coolant System, Class 1 System Leakage Test, Revision 10
WI-PCS, NSSS Walkdown, Revision 0
NDT-PT-01, Procedure Qualification, Revision 2
NDT-PT-02, Procedure Qualification, Revision 0
EM-09-20, Boric Acid Corrosion Control Program, Revision 0
54-ISI-100-14, Remote Ultrasonic Examination of Reactor Head Penetrations, Revision 14
54-ISI-178-05, Ultrasonic Examination of Control Rod Drive Mechanism and Incore Instrumentation Nozzle Temperbead Weld Repair, Revision 5
54-ISI-137-06, Remote Ultrasonic Examination of Reactor Vessel Head Vent Line Penetrations, Revision 6

54-ISI-460, Multi-Frequency Eddy Current Examination of Nozzle Welds and Regions, Revision 2
NDT-VT-09, Bare Metal Visual Examination, Revision 3
SG-SGDA-05-31, Palisades Nuclear Plant REFOUT 17 Outage Condition Monitoring Report and Cycle 18 Operational Assessment, March 2005
PDI Piping and Bolting Program, Krautkramer Model USN-58L and USN-58R, October 2, 2003
PDI Document, Krautkramer Model USN-58L and USN-58R, October 3, 2003
QA-6, Qualification and Certification of NDE and Visual Examination Personnel, Revision 34
QA-37, Qualification of Nondestructive Examination Personnel for Ultrasonic Equipment, Revision 7
54-ISI-24, Written Practice for Personnel Qualification in Eddy Current Examination, Revision 29
WPD9.2, Qualification and Certification of Personnel in Nondestructive Examination, Revision 7
54-ISI-30, Written Practice for the Qualification and Certification of NDE Personnel, Revision 3
WR299696, Replacement of Line Loop-1A Check Valve, October 1, 2004
WR24323635, HPSI Pump P-66B Miniflow Check Valve, October 25, 2004
AR01024003, NDE Examiner did not follow ISI UT Procedure, April 12, 2006
AR01023798, Extra Dose received due to ISI Components not Ready for Exam, April 12, 2006
CAP04414, Steam Generator ET Inspection Indicates Tube Plugged for Wear, September 3, 2004
CAP044215, SG E-50A ET Inspection Identifies Axial Indication in Non-Expanded Tube, October 2, 2004
CAP044293, Foreign Material Found in Secondary Side of Steam Generator E-50B Hotleg, October 4, 2004
CAP034104, Boric Acid Buildup on CV-1059 (Pressurizer Spray Valve From Loop 2A), March 17, 2003
CAP044595, CK-CVC2114 UT Possibly Performed with Flow Through Valve, October 13, 2004
CAP044885, Discrepancies Discovered During VT-2 Level III, Examination Reviews, October 28, 2004
AR 01009004, BACCP Containment Walkdown, December 31, 2005
FP-PE-B31-P8P8-GTSM-037, Groove Welds and Fillet Welds, P8-P8, GTAW/SMAW, Without PWHT, Revision 0
Drawing 5038702E, Palisades CRDM Nozzle ID temper Bead Weld Repair, Revision 4

1R11 Licensed Operator Requalification
Simulator Exercise Guide PL-OPS-SPE-061E, Course N00320, Revision 0

1R13 Maintenance Risk Assessments and Emergent Regulatory Guide Work Evaluation
GOP-14 Shutdown Risk Assessment for Planned Orange Path at Midloop, April 7, 2006
GOP-14 Shutdown Risk Assessment for Planned Orange Path at Midloop, April 23, 2006
GOP-14 Shutdown Risk Assessment for Planned Yellow Path, May 13, 2006
Daily Maintenance Work Week Schedule, May 21-27, June 11-17, June 18-24, 2006

1R14 Operator Performance During Nonroutine Evolutions and Events
SOP1B, Primary Coolant System - Cooldown, Revision 4
Operator Logs, May 10, 2006
WO00284180, CRD-33 Relatch, May 13, 2006
RFL-R-11, Couple CRDMs, Revision 0

Action Plan: Unexpected Flux Tilt During Plant Start-up Following RO-18 Troubleshooting, May 11, 2006
Action Plan: EDG 1-2 Failure to Meet Start Time, June 22, 2006

1R15 Operability Evaluations
OPR01021246-01, CCW Room EQ Issues, Revision 0
FP-OP-OL-1, Corporate Office Quality Procedures: Operability Determination, Revision 1
OPR01023662, Critical Service Water Outlet Piping Downstream of CV-0823 SW Outlet from E-54A Component Cooling Heat Exchanger, Revision 0

1R19 Post Maintenance Testing
WO 00280015, L-1; Disengage from L-6, April 22, 2006
RO-19, Control Rod Position Verification (Rod 26), Revision 21
SOP 6, Reactor Control System, Revision 24
MO-3081, Diagnostic Test Data, April 12, 2006
MO-3081, Diagnostic Test Data, April 5, 2003
RT-71L, TS ADMIN 5.5.2 Pressure Test of ESS Suction Piping, Revision 13
MO-7A-2, Emergent Regulatory Guide Diesel Generator 1-2, Revision 59 and Revision 60
MSM-—57, Universal Diagnostic System Operating Procedure for CV-3070, Completed May 1, 2006
QO-14, Inservice Test Procedure - Service Water Pumps, Revision 24
WO 00157309-02, Auxiliary Feedwater Flow Control to E-50B, June 14, 2006

1R20 Refueling and Other Outage Activities
GOP-9, Plant Cooldown from Hot Standby/Shutdown, Revision 26
Ltr Wiggins to Steffler, Refueling Boron Concentration, March 23, 2006
GOP-8, Power Reduction and Plant Shutdown to Mode 2 or Mode 3 > 525F, Revision 25
GOP-14, Shutdown Cooling Operations, Revision 23
EA-EC-7368-01, Calculation for Radiological Consequences of a Beyond Design Basis Reactor Vessel Head Drop, Revision 0
RO18 Schedule for LLRTs, March 29, 2006 and April 7, 2006
USFAR Chapter 6, Engineered Safeguards Systems (Generic Letter 88-17 Commitments), Revision 24
NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, December 1991
Palisades Response to Generic Letter 88-17, Loss of Decay Heat Removal (60 Day Response), January 3, 1989
RFL-SG-2, S/G Primary Nozzle Dam Installation and Removal, Revision 0
ONP-17, Loss Of Shutdown Cooling, Revision 32
RFL-D-3, Open Equipment Hatch, Revision 0
GOP-11, Refueling Operations and Fuel Handling, Revision 40
Reactor Core Reconfiguration Sequence, Revision 1
RFL-V-7, Fuel Movement, Revision 0
SOP-28, Fuel Handling System, Revision 34

SOP-1C, Primary Coolant System - Heatup, Revision 3
GOP-2, Mode 5 to Mode 3 > or = 525F, Revision 3

1R22 Surveillance Testing
RT-8D, Engineered Safeguards System- Right Channel, Revision 22
RO-105, Full Flow Test for SIT Check Valves and PCS Loop Check Valves, Revision 8
RT-129, Functional Test of Bus 1C Undervoltage Relays, Revision 5
RO-32-47, LLRT - Local Leak Rate Test for Penetration MZ-47, Revision 26
RT-92, Inspection of ECCS Train Containment Sump Suction Inlet, Revision 2

1R23 Temporary Modifications
EC 544, Temporary Jumper of Test Switch Contacts on TPS-187Z-107

2OS1 Access Control to Radiologically Significant Areas: and
2OS2 ALARA Planning and Controls

FP-RP-DP-01, Dosimetry Program, Revision 0
Administrative Procedure 7.13, Radiological Area Access, Revision 11
AR 01024675, Radiography performed without primary/secondary dosimetry
WI-RSD-H-003, Control of Radiography
AR 01024675, Movement of Locked High Radiation Area Boundary Swing Gate
AR 0102302, Crane controls not properly secured and controlled located in the Auxiliary Building Clean Waste Filter Transfer Area
QF-1203, Radiological Work Assessment Form; Work Order/Task No. 26447, approved March 31, 2006
QF-1204, Radiological Work Assessment Form Contamination Control, Work Order 26447, approved March 31, 2006
QF-1205, Radiological Work Assessment Form Exposure Control, Work Order 26447; approved March 31, 2006
QF-1206, Radiological Work Assessment Form Internal Exposure Control, Work Order 26447, approved March 31, 2006
QF-1207, Radiological Planning Checklist, Work Order 26447, approved March 31, 2006
QF-1209, Radiological Pre-Job Briefing Form, Work Order 26447, no date provided
RWP 583, RX Head Insp - Under Reactor Head Entries, March 30, 2006
QF-1203, Radiological Work Assessment Form, Work Order/Task No. 26447, approved March 31, 2006
QF-1204, Radiological Work Assessment Form Contamination Control, Work Order 25806, approved March 31, 2006
QF-1205, Radiological Work Assessment Form Exposure Control, Work Order 25806; approved March 31, 2006
QF-1206, Radiological Work Assessment Form Internal Exposure Control, Work Order 25806, approved March 31, 2006
QF-1207, Radiological Planning Checklist, Work Order 25806, approved March 31, 2006
QF-1209, Radiological Pre-Job Briefing Form; Work Order 25806, no date provided
RWP 597, RX Head Disassembly and Reassembly Maintenance, March 27, 2006

4OA2 Problem Identification and Resolution
WO00026859 02, REFOUT PMs on L-1, L-3, L-6 and L906, April 12, 2006
Operations Department Monthly Performance Report, March 2006

Site DRUM Report for the 1st Quarter of 2006
Monthly Performance Indicators Report, May 2006
Corrective Action Program Performance Indicator Summary, May 2006
FG-PA-CTC-01, CAP Trend Code Manual, Revision 5
Searched on trend activity codes MI (material issue and return), MT (monitoring & trending), PR (procurement) and TV (training development) to assemble list of related action requests

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document and Management System
AFW	Auxiliary Feedwater Pump
ALARA	As Low As Is Reasonably Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DRS	Division of Reactor Safety
EC	Eddy Current
ET	Eddy Current Examination
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FME	Foreign Material Exclusion
FSAR	Final Safety Analysis Report
HPSI	High Pressure Safety Injection
HRA	High Radiation Areas
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
IST	Inservice Test
LCO	Limiting Condition for Operation
MSSV	Main Steam Safety Valve
NCV	Non-Cited Violation
NDE	Nondestructive Examination
OPR	Operability Recommendations
OWA	Operator Work Around
PARS	Publicly Available Records
PCS	Primary Coolant System
PSA	Probabilistic Safety Assessment
RCA	Radiologically Controlled Area
RI	Resident Inspector
RI-ISI	Risk Informed Inservice Inspection
RP	Radiation Protection
RPVCH	Reactor Pressure Vessel Closure Head
RWP	Radiation Work Permits
RVUHP	Reactor Vessel Upper Head Penetration
SDCHX	Shutdown Cooling Heat Exchanger
SEP	Systematic Evaluation Program
SG	Steam Generator
SSC	Structures, Systems, and Components
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Examination
VT	Visual Examination